NEUTRONIC CALCULATIONS IN CORE CONVERSION OF THE IAN-R1 RESEARCH REACTOR FROM MTR HEU TO TRIGA LEU FUEL .

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ABSTRACT

With cooperation of the International Atomic Energy Agency (IAEA), neutronic calculations were carried out for conversion of the IAN-R1 Reactor from MTR-HEU fuel to TRIGA-LEU fuel. In order to establish a staff for neutronic calculation at the Instituto de Ciencias Nucleares y Energías Alternativas (INEA) a program was established. This program included training, acquisition of hardware, software and calculation for the core with MTR-HEU fuel , enriched nominally to 93% and calculation for several arrangements with the TRIGA-LEU fuel, enriched to 19.7%. The results were verified and compared with several groups of calculation at the Instituto Nacional de Investigaciones Nucleares (ININ) in Mexico, and General Atomics (GA) in United States. As a result of this program, several technical reports have been wrote.

1. Introduction.

IAN-R1 is a research reactor pool type which was initially fueled with MTR–HEU enriched to 93% U-235 [1], operated since 1965 to 10 kW (t) and was upgraded to 30 kW(t) in 1980. General Atomics (GA) achieved in 1997 the conversion of HEU fuel to LEU fuel TRIGA ($UzrH_{1.6}$) type, and upgraded the reactor power to 100 kW(t) [2]. This paper describes the steps carried out by staff of IAN-R1 research reactor in order to evaluate the new core fueled with TRIGA LEU, establishing comparisons between MTR-HEU and TRIGA-LEU core, models and calculation methods used by GA and staff of IAN-R1.

To compare the general performance of both cores HEU and LEU, initially neutronic calculations were achieved for a neutron energy structure of two groups with thermal (<0.625 eV) and fast (>0.625 eV). The TRIGA LEU core evaluated in this paper was taken from the first proposal established by GA to interchange the older HEU plate-type [3,4,5,6]. Core configurations, unit cells, neutron cross sections and fluxes, are mainly the parameters that have been compared. In addition this paper also describes the different neutron energy structures that have been considered, the different codes, and their influence in neutron cross sections for the TRIGA LEU core.

2. Preliminary Analysis

2.1. MTR Core

The HEU core is a flat-plate design and it consists of a 4x4 array of fuel elements surrounded by 20 graphite reflector elements, and six of the graphite elements contain circular irradiation spaces. The core contains 13 standard fuel elements, 10 fuel plates/element, 3 of the 16 fuel assemblies are control elements and 6 fuel plates/control element [1]. The cooling-moderating material is light water, and the fuel elements are reflected on four sides with graphite and on the top and bottom sides with water. Fig 1., shows the core configuration with MTR-HEU fuel.

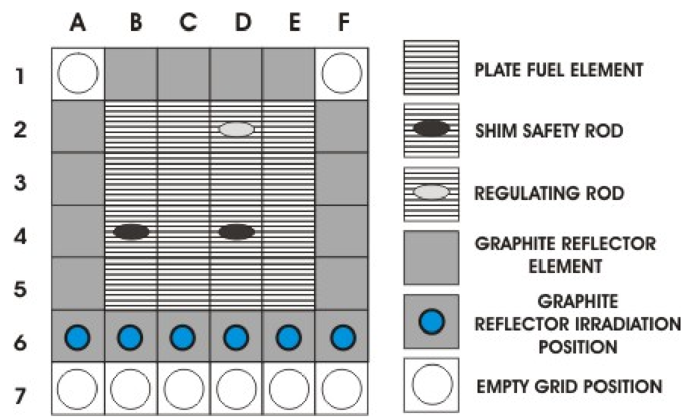

Fig. 1. Core configuration with MTR-HEU fuel.

The description of slab geometry used in unit cell for the generation of thermal and fast cross sections for the MTR-HEU core is given in Table 1 [7].

Table 1. Unit Cell Description

Region	Distance (cm)	Nuclide	wt %
Meat	0.0254	U	0.26753
		U-235	0.240777
		U-238	0.026753
		Al	0.73247
Al	0.9245	Al	1.00000
Water	0.385443	O	0.8881056
		H	0.1118944
Al	0.417203	Al	1.00000
Water	0.4769433	O	0.8881056
		H	0.1118944

2.2. TRIGA Core.

The LEU core has 8 fuel cluster TRIGA standard, 3 cluster each one with 3 fuel rods and a guide tube for the control rods and 3 cluster each one with 3 fuel rods and a guide tube as an irradiation tube. The radial reflector consists of twenty graphite elements, six of which are used for isotope production, the top and bottom reflectors are the cylindrical graphite installed above and below of the active fuel section in each fuel rod [4]. The core is cooled by natural convection flow around the fuel elements. Fig. 2., shows the core configuration with TRIGA-LEU fuel.

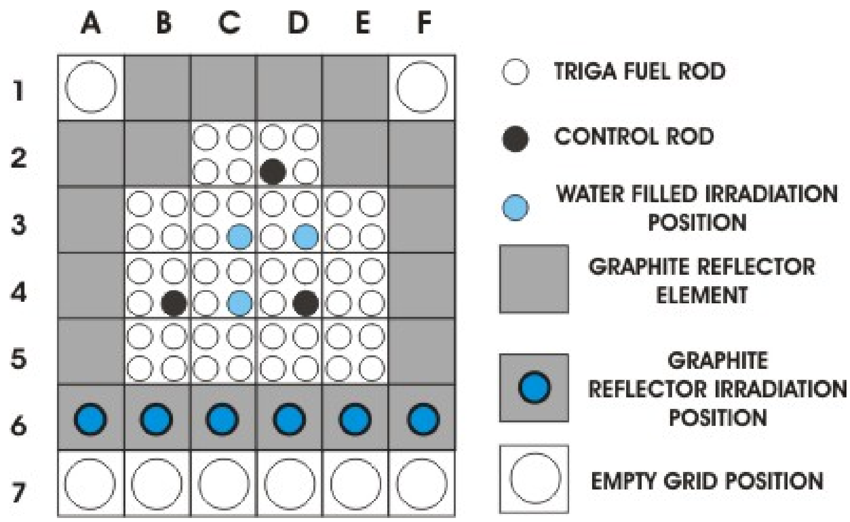

Fig.2. Core configuration with TRIGA-LEU fuel.

The unit cell description used in the generation of thermal and fast cross sections for the TRIGA core is given in Table 2 [8].

Table 2. Unit Cell Description

Region	Radius (cm)	Nuclide	Atoms/b-cm
Zr	0.3150	Zr	0.042918
ZrH	1.6929	H	0.054709
		Zr	0.034191
		U-235	0.00036369
		U-238	0.0014591
Cladding	1.7462	SS	0.0870
Water	2.1746	H	0.066742
		O	0.033379

2.3. Calculation Methods.

In this analysis neutron cross sections are generated for two neutron energy groups. All neutron cross sections for thermal and fast energies are generated using the WIMS code[9]. Absorptions and yields cross-sections were obtained for thermal (< 0.625 eV) and fast (>0.625 eV) energies. The neutron energy group structure and cross sections are presented in Table 3 [7].

Table 3. Neutron cross sections for MTR-HEU and TRIGA-LEU

Energy Interval (eV)	Absorption		Fission yields	
	MTR-HEU	TRIGA-LEU	MTR-HEU	TRIGA-LEU
$1.0\text{x}\ 10^{7}$ - 0.625	$2.08900\text{x}10^{-3}$	$6.59875\text{x}10^{-3}$	$2.52466\text{x}10^{-3}$	$5.07469\text{x}10^{-3}$
0.625 - 0.0	$6.56038\text{x}10^{-2}$	$1.06613\text{x}10^{-1}$	$1.03453\text{x}10^{-1}$	$1.70662\text{x}10^{-1}$

For the determination of axial buckling, first was obtained the reactivity using a three-dimensional calculation and after the values of the bucklings were adjusted with a two-dimensional calculation. The axial buckling for MTR-HEU obtained is $B^2_z = 1.80 \times 10^{-3}$ cm^2, and for TRIGA-LEU $B^2_z = 3.76 \times 10^{-3}$ cm^2. Two and three dimensional calculations were done using the diffusion theory code CITATION [10].

From diffusion code CITATION thermal neutron fluxes were obtained for graphite reflectors irradiation on mid-plane, and axial flux for the graphite reflector irradiation located in C-6 as shown Fig.1, and Fig.2. They were plotted in figures 3.1 and 3.2.

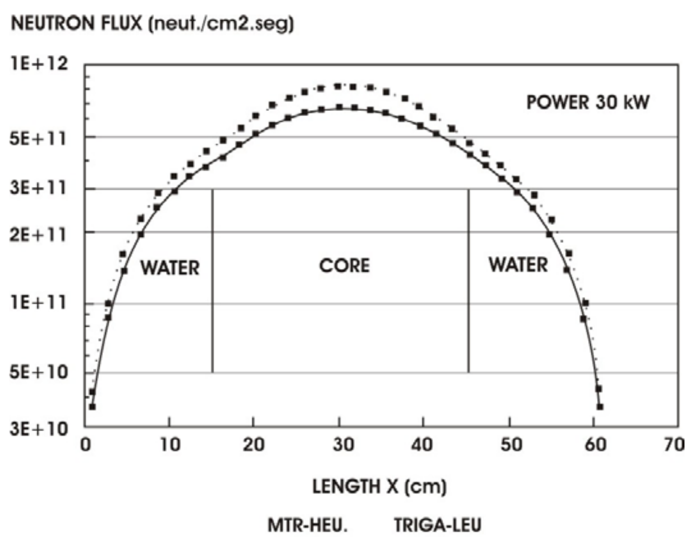

Fig.3.1. Mid-plane thermal neutron flux in graphite reflectors irradiation

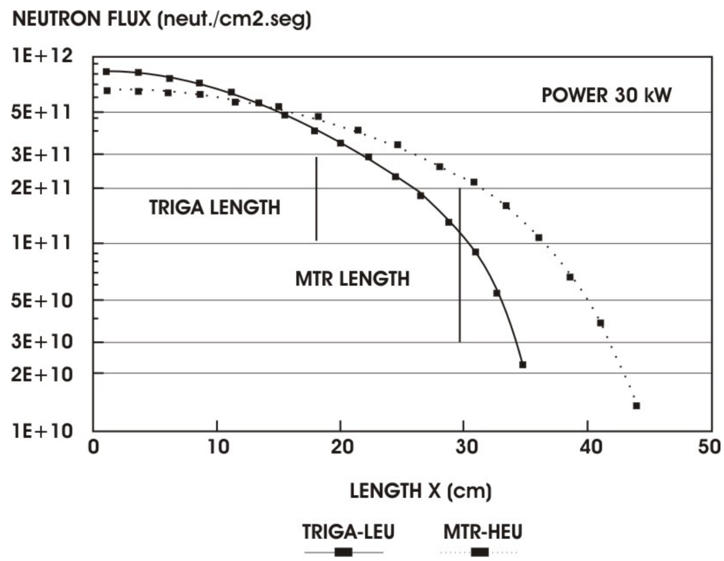

Fig. 3.2. Axial thermal neutron flux in 6-C graphite reflector irradiation

3. Subsequent Analysis.

As GA used slightly different values of radius and atoms/b-cm for neutronic calculation[11], and used the neutron energy group structure of the Table 4., new values were calculated for absorptions and yields cross-section with the WIMS code.

Table 4. Neutron Energy Group Structure

Group	Energy Interval (eV)		
1	1.49×10^7	-	6.08×10^5
2	6.08×10^5	-	9.12×10^3
3	9.12×10^3	-	1.125
4	1.125	-	0.420
5	0.420	-	0.140
6	0.140	-	0.050
7	0.050	-	0.002

According to the new values of GA for radius, atoms/b-cm, thermal and fast regions for neutron energies, absorptions and yields cross section were calculated for a new energy group structure with thermal (< 1.125) and fast neutron (>1.125 eV), and same time cross sections with the energy intervals of the table 3. In Table 5. are gave the values for cross sections and the effective multiplication factor Keff calculated by GA [12] and INEA.

Table 5. Neutron cross sections and Keff for TRIGA-LEU

Energy Interval (eV)		Code	Absorption	Fission yields	Keff
$1.49x10^{7}$	1.125	GGC-5	$6.274560x10^{-03}$	$4.219266x10^{-03}$	
1.125	0.002	GGF	$1.003339x10^{-01}$	$1.601236x10^{-01}$	
		2DB			1.002393
$1.0x10^{7}$	1.125	WIMS	$6.19720x10^{-03}$	$4.46617x10^{-03}$	
1.125	0.000	WIMS	$1.00186x10^{-01}$	$1.61148x10^{-01}$	
		CITATION			1.009221
$1.0x10^{7}$	0.625	WIMS	$6.49589x10^{-03}$	$5.00536x10^{-03}$	
0.625	0.000	WIMS	$1.04932x10^{-01}$	$1.68834x10^{-01}$	
		CITATION			1.009408

4. Conclusions.

The neutronic calculations carried out by staff of IAN-R1 using WIMS and CITATION codes let us conclude that thermal neutron flux for TRIGA-LEU core in graphite reflectors irradiation is similar to the MTR-HEU core. To neutron energy group structure maintained to two groups and adjusting the boundary of the thermal region to 1.125, the cross sections and the effective multiplication factor obtained by staff of IAN-R1 tend to values of GA.

5. Acknowledgments.

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6. References.

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